NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	9	71-9235	USA/9235/B(U)F-96	1	OF	12

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc., application dated March 1, 2004, as supplemented

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: NAC-STC

(2) Description: For descriptive purposes, all dimensions are approximate nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

A steel, lead and polymer (NS4FR) shielded shipping cask for (a) directly loaded irradiated PWR fuel assemblies, (b) intact, damaged and/or the fuel debris of Yankee Class or Connecticut Yankee irradiated PWR fuel assemblies in a canister, and (c) non-fissile, solid radioactive materials (referred to hereafter as Greater Than Class C (GTCC) as defined in 10 CFR Part 61) waste in a canister. The cask body is a right circular cylinder with an impact limiter at each end. The package has approximate dimensions as follows:

Cavity diameter	71 inches
Cavity length	165 inches
Cask body outer diameter	87 inches
Neutron shield outer diameter	99 inches
Lead shield thickness	3.7 inches
Neutron shield thickness	5.5 inches
Impact limiter diameter	124 inches
Package length:	
without impact limiters	193 inches
with impact limiters	257 inches

The maximum gross weight of the package is about 260,000 lbs.

The cask body is made of two concentric stainless steel shells. The inner shell is 1.5 inches thick and has an inside diameter of 71 inches. The outer shell is 2.65 inches thick and has

5.(a)(2) Description (Continued)

an outside diameter of 86.7 inches. The annulus between the inner and outer shells is filled with lead.

The inner and outer shells are welded to steel forgings at the top and bottom ends of the cask. The bottom end of the cask consists of two stainless steel circular plates which are welded to the bottom end forging. The inner bottom plate is 6.2 inches thick and the outer bottom plate is 5.45 inches thick. The space between the two bottom plates is filled with a 2-inch thick disk of a synthetic polymer (NS4FR) neutron shielding material.

The cask is closed by two steel lids which are bolted to the upper end forging. The inner lid (containment boundary) is 9 inches thick and is made of Type 304 stainless steel. The outer lid is 5.25 inches thick and is made of SA-705 Type 630, H1150 or 17-4PH stainless steel. The inner lid is fastened by 42, 1-1/2-inch diameter bolts and the outer lid is fastened by 36, 1-inch diameter bolts. The inner lid is sealed by two O-ring seals. The outer lid is equipped with a single O-ring seal. The inner lid is fitted with a vent and drain port which are sealed by O-rings and cover plates. The containment system seals may be metallic or Viton. Viton seals are used only for directly-loaded fuel that is to be shipped without long-term interim storage.

The cask body is surrounded by a 1/4-inch thick jacket shell constructed of 24 stainless steel plates. The jacket shell is 99 inches in diameter and is supported by 24 longitudinal stainless steel fins which are connected to the outer shell of the cask body. Copper plates are bonded to the fins. The space between the fins is filled with NS4FR shielding material.

Four lifting trunnions are welded to the top end forging. The package is shipped in a horizontal orientation and is supported by a cradle under the top forging and by two trunnion sockets located near the bottom end of the cask.

The package is equipped at each end with an impact limiter made of redwood and balsa. Two impact limiter designs consisting of a combination of redwood and balsa wood, encased in Type 304 stainless steel are provided to limit the g-loads acting on the cask during an accident. The predominately balsa wood impact limiter is designed for use with all the proposed contents. The predominately redwood impact limiters may only be used with directly loaded fuel or the Yankee-MPC configuration.

The contents are transported either directly loaded (uncanistered) into a stainless steel fuel basket or within a stainless steel transportable storage canister (TSC).

The directly loaded fuel basket within the cask cavity can accommodate up to 26 PWR fuel assemblies. The fuel assemblies are positioned within square sleeves made of stainless steel. Boral or TalBor sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 31, ½-inch thick, 71-inch diameter stainless steel disks. The basket also has 20 heat transfer disks made of Type 6061-T651 aluminum alloy. The support disks

5.(a)(2) Description (Continued)

and heat transfer disks are connected by six, 1-5/8-inch diameter by 161-inch long threaded rods made of Type 17-4 PH stainless steel.

The TSC shell, bottom plate, and welded shield and structural lids are fabricated from stainless steel. The bottom is a 1-inch thick steel plate for the Yankee-MPC and 1.75-inch thick steel plate for the CY-MPC. The shell is constructed of 5/8-inch thick rolled steel plate and is 70 inches in diameter. The shield lid is a 5-inch thick steel plate and contains drain and fill penetrations for the canister. The structural lid is a 3-inch thick steel plate. The canister contains a stainless steel fuel basket that can accommodate up to 36 intact Yankee Class fuel assemblies and Reconfigured Fuel Assemblies (RFAs), or up to 26 intact Connecticut Yankee fuel assemblies with RFAs, with a maximum weight limit of 35,100 lbs. Alternatively, a stainless steel GTCC waste basket is used for up to 24 containers of waste.

One TSC fuel basket configuration can store up to 36 intact Yankee Class fuel assemblies or up to 36 RFAs within square sleeves made of stainless steel. Boral sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 22 ½-inch thick , 69-inch diameter stainless steel disks, which are spaced about 4 inches apart. The support disks are retained by split spacers on eight 1.125-inch diameter stainless steel tie rods. The basket also has 14 heat transfer disks made of Type 6061-T651 aluminum alloy.

The second fuel basket is designed to store up to 26 Connecticut Yankee Zirc-clad assemblies enriched to 3.93 wt. percent, stainless steel clad assemblies enriched up to 4.03 wt. percent, RFAs, or damaged fuel in CY-MPC damaged fuel cans (DFCs). Zirc-clad fuel enriched to between 3.93 and 4.61 wt. percent, such as Westinghouse Vantage 5H fuel, must be stored in the 24-assembly basket. Assemblies approved for transport in the 26-assembly configuration may also be shipped in the 24-assembly configuration. The construction of the two basket configurations is identical except that two fuel loading positions of the 26-assembly basket are blocked to form the 24-assembly basket.

RFAs can accommodate up to 64 Yankee Class fuel rods or up to 100 Connecticut Yankee fuel rods, as intact or damaged fuel or fuel debris, in an 8x8 or 10x10 array of stainless steel tubes, respectively. Intact and damaged Yankee Class or Connecticut Yankee fuel rods, as well as fuel debris, are held in the fuel tubes. The RFAs have the same external dimensions as a standard intact Yankee Class, or Connecticut Yankee fuel assembly.

The TSC GTCC basket positions up to 24 Yankee Class or Connecticut Yankee waste containers within square stainless steel sleeves. The Yankee Class basket is supported laterally by eight 1-inch thick, 69-inch diameter stainless steel disks. The Yankee Class basket sleeves are supported full-length by 2.5-inch thick stainless steel support walls. The support disks are welded into position at the support walls. The Connecticut Yankee GTCC basket is a right-circular cylinder formed by a series of 1.75-inch thick Type 304 stainless steel plates, laterally supported by 12 equally spaced welded 1.25-inch thick Type 304 stainless steel outer ribs. The GTCC waste containers accommodate radiation activated and surface contaminated steel, cutting debris (dross) or filter media, and have the same external dimensions of Yankee Class or Connecticut Yankee fuel assemblies.

5.(a)(2) Description (Continued)

The Yankee Class TSC is axially positioned in the cask cavity by two aluminum honeycomb spacers. The spacers, which are enclosed in a Type 6061-T651 aluminum alloy shell, position the canister within the cask during normal conditions of transport. The bottom spacer is 14-inches high and 70-inches in diameter, and the top spacer is 28-inches high and also 70-inches in diameter.

The Connecticut Yankee TSC is axially positioned in the cask cavity by one stainless steel spacer located in the bottom of the cask cavity.

5.(a)(3) Drawings

(i) The cask is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-800, sheets 1-3, Rev. 14
423-802, sheets 1-7, Rev. 20
423-803, sheets 1-2, Rev. 8
423-804, sheets 1-3, Rev. 8
423-805, sheets 1-2, Rev. 6
423-806, Rev. 7
423-807, sheets 1-3, Rev. 3
423-811, sheets 1-2, Rev. 11
423-812, Rev. 6
423-900, Rev. 6
423-209, Rev. 0
423-210, Rev. 0
423-901, Rev. 2

(ii) For the directly loaded configuration, the basket is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-870, Rev. 5
423-871, Rev. 5
423-872, Rev. 6
423-873, Rev. 2
423-874, Rev. 2
423-875, sheets 1-2, Rev. 7

(iii) For the Yankee Class TSC configuration, the canister, and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

455-800, sheets 1-2, Rev. 2
455-801, sheets 1-2, Rev. 3
455-820, sheets 1-2, Rev. 2
455-870, Rev. 5
455-871, sheets 1-2, Rev. 8
455-871, sheets 1-3, Rev. 7P2[1]
455-872, sheets 1-2, Rev. 12
455-872, sheets 1-2, Rev. 11P1[1]
455-873, Rev. 4
455-881, sheets 1-3, Rev. 8
455-887, sheets 1-3, Rev. 4
455-888, sheets 1-2, Rev. 8
455-891, sheets 1-2, Rev. 1
455-891, sheets 1-3, Rev. 2PO[1]
455-892, sheets 1-2, Rev. 3
455-892, sheets 1-3, Rev. 3P0[1]
455-893, Rev. 3
455-894, Rev. 2
455-895, sheets 1-2, Rev. 5
455-895, sheets 1-2, Rev. 5P0[1]
455-901, Rev. 0P0[1]
455-902, sheets 1-5, Rev. 0P4[1]
455-919, Rev. 2

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[1]Drawing defines the alternate configuration that accommodates the Yankee-MPC damaged fuel can.

5.(a)(3) Drawings (Continued)

(iv) For the Yankee Class TSC configuration, RFAs are constructed and assembled in accordance with the following Yankee Atomic Electric Company Drawing Nos.:

YR-00-060, Rev. D3
YR-00-061, Rev. D4
YR-00-062, sheet 1, Rev. D4
YR-00-062, sheet 2, Rev. D2
YR-00-062, sheet 3, Rev. D1
YR-00-063, Rev. D4
YR-00-064, Rev. D4
YR-00-065, Rev. D2
YR-00-066, sheet 1, Rev. D5
YR-00-066, sheet 2, Rev. D3

(v) The Balsa Impact Limiters are constructed and assembled in accordance with the following NAC International Drawing Nos.:

423-257, Rev. 2
423-258, Rev. 2
423-843, Rev. 2
423-859, Rev. 0

(vi) For the Connecticut Yankee TSC configuration, the canister and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-801, sheets, 1-2, Rev. 1
414-820, Rev. 0
414-870, Rev. 3
414-871, sheets 1-2, Rev. 6
414-872, sheets 1-3, Rev. 6
414-873, Rev. 2
414-874, Rev. 0
414-875, Rev. 0
414-881, sheets 1-2, Rev. 4
414-882, sheets 1-2, Rev. 4
414-887, sheets 1-4, Rev. 4
414-888, sheets 1-2, Rev. 4
414-889, sheets 1-3, Rev. 7
414-891, Rev. 3
414-892, sheets 1-3, Rev. 3
414-893, sheets,1-2, Rev. 2
414-894, Rev. 0
414-895, sheets 1-2, Rev. 4

(vii) For the Connecticut Yankee TSC configuration, DFCs and RFAs are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-901, Rev. 1
414-902, sheets 1-3, Rev. 3
414-903, sheets 1-2, Rev. 1
414-904, sheets 1-3, Rev. 0

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies with uranium oxide pellets. Each fuel assembly may have a maximum burnup of 45 GWD/MTU. The minimum fuel cool time is defined in the Fuel Cool Time Table, below. The maximum heat load per assembly is 850 watts. Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	14x14	15x15	16x16	17x17	17x17 (OFA)	Framatome-Cogema 17x17
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirconium Alloy
Maximum Initial Uranium Content (kg/assembly)	407	469	402.5	464	426	464
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2	4.2	4.5
Minimum Initial Enrichment (wt% ^{235}U)	1.7	1.7	1.7	1.7	1.7	1.7
Assembly Cross-Section (inches)	7.76 to 8.11	8.20 to 8.54	8.10 to 8.14	8.43 to 8.54	8.43	8.425 to 8.518
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264	264[1]
Fuel Rod OD (inch)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360	0.3714 to 0.3740
Minimum Cladding Thickness (inch)	0.023	0.024	0.025	0.023	0.023	0.0204
Pellet Diameter (inch)	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088	0.3224 to 0.3230
Maximum Active Fuel Length (inches)	146	144	137	144	144	144.25

Notes:
[1] - Fuel rod positions may also be occupied by solid poison shim rods or solid zirconium alloy or stainless steel fill rods.

5.(b)(1)(i) Contents - Type and Form of Material - Irradiated PWR fuel assemblies (Continued)

FUEL COOL TIME TABLE
Minimum Fuel Cool Time in Years

	Fuel Assembly Burnup (BU)															
Uranium Enrichment (wt% U-235)	BU $\le$ 30 GWD/MTU				30 < BU $\le$ 35 GWD/MTU				35 < BU $\le$ 40 GWD/MTU				40 < BU $\le$ 45 GWD/MTU			
Fuel Type	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.7$\le$E<1.9	8	7	6	7	10	10	7	9	--	--	--	--	--	--	--	--
1.9$\le$E<2.1	7	7	5	7	9	9	7	8	12	13	9	11	--	--	--	--
2.1$\le$E<2.3	7	7	5	6	9	8	6	8	11	11	8	10	--	--	--	--
2.3$\le$E<2.5	6	6	5	6	8	8	6	7	10	10	8	9	14	15	12	14
2.5$\le$E<2.7	6	6	5	6	8	7	6	7	10	9	7	9	13	14	10	12
2.7$\le$E<2.9	6	6	5	5	7	7	5	6	9	9	7	8	12	12	9	11
2.9$\le$E<3.1	6	5	5	5	7	7	5	6	9	8	6	8	11	11	8	10
3.1$\le$E<3.3	5	5	5	5	7	6	5	6	8	8	6	7	10	10	8	9
3.3$\le$E<3.5	5	5	5	5	6	6	5	6	8	7	6	7	10	10	7	9
3.5$\le$E<3.7	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.7$\le$E<3.9	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.9$\le$E<4.1	5	5	5	5	6	6	5	6	7	7	6	7	8	9	7	9
4.1$\le$E$\le$4.2	5	5	5	5	5	6	5	6	6	7	6	7	8	8	7	9
4.2<E<4.3	--	--	--	5[1]	--	--	--	6[1]	--	--	--	7[1]	--	--	--	9[1]
4.3$\le$E$\le$4.5	--	--	--	5[1]	--	--	--	6[1]	--	--	--	7[1]	--	--	--	8[1]

Notes:
[1] - Framatome-Cogema 17x17 fuel only.

5.(b)(1) Contents - Type and Form of Material (Continued)

(ii) Irradiated intact Yankee Class PWR fuel assemblies or RFAs within the TSC. The maximum initial fuel pin pressure is 315 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	UN 16x16	CE [1] 16x16	West. 18x18	Exxon [2] 16x16	Yankee RFA	Yankee DFC
Cladding Material	Zircaloy	Zircaloy	SS	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Rods per Assembly	237	231	305	231	64	305
Maximum Initial Uranium Content (kg/assembly)	246	240	287	240	70	287
Maximum Initial Enrichment (wt% ^{235}U)	4.0	3.9	4.94	4.0	4.94	4.97 [3]
Minimum Initial Enrichment (wt% ^{235}U)	4.0	3.7	4.94	3.5	3.5	3.5 [3]
Maximum Assembly Weight (lbs)	≤950	≤950	≤950	≤950	≤950	≤950
Maximum Burnup (Mwd/MTU)	32,000	36,000	32,000	36,000	36,000	36,000
Maximum Decay Heat per Assembly (kW)	0.28	0.347	0.28	0.34	0.11	0.347
Minimum Cool Time (yrs)	11.0	8.1	22.0	10.0	8.0	8.0
Maximum Active Fuel Length (in)	91	91	92	91	92	N/A

Notes:

1. Combustion Engineering (CE) fuel with a maximum burnup of 32,000 Mwd/MTU, a minimum enrichment of 3.5 wt. percent ^{235}U, a minimum cool time of 8.0 years, and a maximum decay heat per assembly of 0.304 kW is authorized.
2. Exxon assemblies with stainless steel in-core hardware shall be cooled a minimum of 16.0 years with a maximum decay heat per assembly of 0.269 kW.
3. Stated enrichments are nominal values (fabrication tolerances are not included).

5.(b)(1) Contents - Type and Form of Material (Continued)

(iii) Solid, irradiated, and contaminated hardware and solid, particulate debris (dross) or filter media placed in a GTCC waste container, provided the quantity of fissile material does not exceed a Type A quantity, and does not exceed the mass limits of 10 CFR 71.15.

(iv) Irradiated intact and damaged Connecticut Yankee (CY) Class PWR fuel assemblies (including optional stainless steel rods inserted into the CY intact and damaged fuel assembly reactor control cluster assembly (RCCA) guide tubes that do not contain RCCAs), RFAs, or DFCs within the TSC. The maximum initial fuel pin pressure is 475 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	PWR [1] 15x15	PWR [2] 15x15	PWR [3]	CY-MPC RFA[4]	CY-MPC DFC[5]
Cladding Material	SS	Zircaloy	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Assemblies	26	26	24	4	4
Maximum Initial Uranium Content (kg/assembly)	433.7	397.1	390	212	433.7
Maximum Initial Enrichment (wt% ^{235}U)	4.03	3.93	4.61	4.61[6]	4.61[6]
Minimum Initial Enrichment (wt% ^{235}U)	3.0	2.95	2.95	2.95	2.95
Maximum Assembly Weight (lbs)	≤1,500	≤1,500	≤1,500	≤1,600	≤1,600
Maximum Burnup (Mwd/MTU)	38,000	43,000	43,000	43,000	43,000
Maximum Decay Heat per Assembly (kW)	0.654	0.654	0.654	0.321	0.654
Minimum Cool Time (yrs)	10.0	10.0	10.0	10.0	10.0
Maximum Active Fuel Length (in)	121.8	121.35	120.6	121.8	121.8

Notes:

1. Stainless steel assemblies manufactured by Westinghouse Electric Co., Babcock & Wilcox Fuel Co., Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co.
2. Zircaloy spent fuel assemblies manufactured by Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co., and Babcock & Wilcox Fuel Co.
3. Westinghouse Vantage 5H zircaloy clad spent fuel assemblies have an initial uranium enrichment > 3.93 % wt. U^{235}.
4. Reconfigured Fuel Assemblies (RFA) must be loaded in one of the 4 oversize fuel loading positions.
5. Damaged Fuel Cans (DFC) must be loaded in one of the 4 oversize fuel loading positions.
6. Enrichment of the fuel within each DFC or RFA is limited to that of the basked configuration in which it is loaded.

5.(b) Contents (Continued)

(2) Maximum quantity of material per package

(i) For the contents described in Item 5.(b)(1)(i): 26 PWR fuel assemblies with a maximum total weight of 39,650 lbs. and a maximum decay heat not to exceed 22.1 kW per package.

(ii) For the contents described in Item 5.(b)(1)(ii): Up to 36 intact fuel assemblies to the maximum content weight limit of 30,600 lbs. with a maximum decay heat of 12.5 kW per package. Intact fuel assemblies shall not contain empty fuel rod positions and any missing rods shall be replaced by a solid Zircaloy or stainless steel rod that displaces an equal amount of water as the original fuel rod. Mixing of intact fuel assembly types is authorized.

(iii) For intact fuel rods, damaged fuel rods and fuel debris of the type described in Item 5.(b)(1)(ii): up to 36 RFAs, each with a maximum equivalent of 64 full length Yankee Class fuel rods and within fuel tubes. Mixing of directly loaded intact assemblies and damaged fuel (within RFAs) is authorized. The total weight of damaged fuel within RFAs or mixed damaged RFA and intact assemblies shall not exceed 30,600 lbs. with a maximum decay heat of 12.5 kW per package.

(iv) For the contents described in Item 5.(b)(1)(iii): for Connecticut Yankee GTCC waste up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 196,000 curies. The total weight of the waste containers shall not exceed 18,743 lbs. with a maximum decay heat of 5.0 kW. For all others, up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 125,000 curies. The total weight of the waste and containers shall not exceed 12,340 lbs. with a maximum decay heat of 2.9 kW.

(v) For the contents described in Item 5.(b)(1)(iv): up to 26 Connecticut Yankee fuel assemblies, RFAs or damaged fuel in CY-MPC DFCs for stainless steel clad assemblies enriched up to 4.03 wt. percent and Zirc-clad assemblies enriched up to 3.93 wt. percent. Westinghouse Vantage 5H fuel and other Zirc-clad assemblies enriched up to 4.61 wt. percent must be installed in the 24-assembly basket, which may also hold other Connecticut Yankee fuel types. The construction of the two basket configurations is identical except that two fuel loading positions of the 26 assembly basket are blocked to form the 24 assembly basket. The total weight of damaged fuel within RFAs or mixed damaged RFAs and intact assemblies shall not exceed 35,100 lbs. with a maximum decay heat of 0.654 kW per assembly for a canister of 26 assemblies. A maximum decay heat of 0.321 kW per assembly for Connecticut Yankee RFAs and of 0.654 kW per canister for the Connecticut Yankee DFCs is authorized.

5.(c) Criticality Safety Index: 0.0

6. Known or suspected damaged fuel assemblies or rods (fuel with cladding defects greater than pin holes and hairline cracks) are not authorized, except as described in Item 5.(b)(2)(iii).

7. For contents placed in a GTCC waste container and described in Item 5.(b)(1)(iii): and which contain organic substances which could radiolytically generate combustible gases, a determination must be made by tests and measurements or by analysis that the following criteria are met over a period of time that is twice the expected shipment time:

   The hydrogen generated must be limited to a molar quantity that would be no more than 4% by volume (or equivalent limits for other inflammable gases) of the TSC gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F). For determinations performed by analysis, the amount of hydrogen generated since the time that the TSC was sealed shall be considered.

8. For damaged fuel rods and fuel debris of the quantity described in Item 5.(b)(2)(iii) and 5.(b)(2)(v): if the total damaged fuel plutonium content of a package is greater than 20 Ci, all damaged fuel shall be enclosed in a TSC which has been leak tested at the time of closure. For the Yankee Class TSC the leak test shall have a test sensitivity of at least 4.0 X 10^{-8} cm^3/sec (helium) and shown to have a leak rate no greater than 8.0 X 10^{-8} cm^3/sec (helium). For the Connecticut Class TSC the leak test shall have a test sensitivity of at least 1.0 X 10^{-7} cm^3/sec (helium) and shown to have a leak rate no greater than 2.0 X 10^{-7} cm^3/sec (helium).

9. In addition to the requirements of Subpart G of 10 CFR Part 71:

   (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

   (b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented, except that the thermal testing of the package (including the thermal acceptance test and periodic thermal tests) must be performed as described in NAC-STC Safety Analysis Report.

   (c) For packaging Serial Numbers STC-1 and STC-2, only one of these two packagings must be subjected to the thermal acceptance test as described in Section 8.1.6 of the NAC-STC Safety Analysis Report.

10. Prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar and the system used to support and secure the package during transport.

11. Prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.

12. Transport by air is not authorized.

13. Packagings may be marked with Package Identification Number USA/9235/B(U)F-85 until April 30, 2007, and must be marked with Package Identification Number USA/9235/B(U)F-96 after April 30, 2007.

14. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

15. Revision No. 8 of this certificate may be used until April 30, 2007.

16. Expiration date: March 31, 2009.

REFERENCES

NAC International, Inc., application dated: March 1, 2004.

NAC International, Inc., supplements dated: August 4, and November 1, 2005; and March 1, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Robert A. Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: Apr. 25, 2006

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION REPORT

Docket No. 71-9235
Model No. NAC-STC Package
Certificate of Compliance No. 9235
Revision No. 9

SUMMARY

By application dated August 4, 2005, as supplemented on November 1, 2005, and March 1, 2006, NAC International, Inc. (NAC) requested an amendment to Certificate of Compliance No. 9235, for the Model No. NAC-STC package. NAC requested that the package identification number of the package be revised to include the "-96" designation. NAC provided amendment pages and made editorial, administrative, and technical changes throughout the document and drawings.

Based on the statements and representations in the application, as supplemented, and for the reasons stated in this Safety Evaluation Report, the staff agrees that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

NAC requested an amendment to Certificate of Compliance No. 9235 for its Model No. NAC-STC package to include the designation "-96" in the identification number, as specified in 10 CFR 71.19(e). To support its request for the "-96" designation, NAC provided a table addressing the changes to 10 CFR Part 71 (69 FR 3698), and a discussion of how each change has been addressed. The staff evaluated the applicant's request, as described below.

- Issue 1, Changing Part 71 to the International Systems of Units (SI) only.

  This proposal was not adopted into the final rule, and therefore no changes are needed in the package application or the Certificate of Compliance to conform to the new rule.

- Issue 2, Radionuclide Exemption Values.

  The final rule adopted radionuclide activity concentration values and consignment activity limits in TS-R-1 for the exemption from regulatory requirements for the shipment or carriage of certain radioactive low-level materials. In addition, the final rule adopted an exemption from regulatory requirements for certain natural material and ores containing naturally occurring radionuclides. The applicant indicated that this revision was not applicable to the Model No. NAC-STC package. The staff agrees based on the design purpose of the Model No. NAC-STC package and the allowed contents specified in the certificate. Thus, no changes are needed to conform to the new rule.

- Issue 3, Revision of A_1 and A_2.

The final rule adopted changes in the A_1 and A_2 values from TS-R-1, with the exception of two radionuclides. The A_1 and A_2 values were modified in TS-R-1 based on refined modeling of possible doses from radionuclides, and the NRC agreed that incorporating the latest in dosimetric modeling would improve transportation regulations. The applicant revised all SAR sections affected by this change to incorporate these new values. As discussed in Section/Chapter 4 of this SER, the revised A_1 and A_2 values do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

- Issue 4, Uranium Hexafluoride (UF_6) Package Requirements.

The final rule allows uranium hexafluoride packages to be evaluated for criticality safety without considering the in-leakage of water into the containment system provided certain conditions are met, including that the uranium is enriched to not more than 5 weight percent uranium-235. The Model No. NAC-STC package is not authorized for the transport of uranium hexafluoride. Therefore, no changes are needed to conform to the new rule.

- Issue 5, Criticality Safety Index (CSI).

The final rule adopted the CSI requirement from IAEA Transportation Safety Standards (TS-R-1). The applicant revised Chapters 1, 5, and 6 of the application to incorporate the CSI nomenclature.

- Issue 6, Type C Packages and Low Dispersible Material.

This proposal was not adopted for the final rule. Thus, no change is needed to conform to the new rule.

- Issue 7, Deep Immersion Test.

The final rule adopted an extension of the previous version of 10 CFR 71.61 from packages for irradiated fuel to any Type B package containing activity greater than 10^5 A_2. The Model No. NAC-STC package is a Type B(U)-85 package and had been previously shown to meet the deep immersion requirements of 10 CFR 71.61. The extension of this requirement to any package containing an activity higher than 10^5 A_2 does not exclude or prevent the NAC-STC package from meeting this requirement. Therefore, no changes are needed to conform to the new rule.

- Issue 8, Grandfathering Previously Approved Packages.

The final rule adopted a process for allowing continued use, for specific periods of time, of previously approved packaging designs without demonstrating compliance to the final rule. In accordance with 10 CFR 71.19(e), the applicant submitted this information demonstrating compliance with the final rule. Thus, grandfathering of the Model No. NAC-STC package is not necessary.

- Issue 9, Changes to Various Definitions.

  The final rule adopted several revised and new definitions. These changes were adopted to provide clarity to Part 71. A definition of "Criticality Safety Index" (CSI) has been added and "Criticality Safety Index" has been substituted for "Transportation Index for nuclear criticality control" throughout the application.

- Issue 10, Crush Test for Fissile Material Packages.

  The revised 10 CFR 71.73 expanded the applicability of the crush test to fissile material packages. The crush test is required for packages with a mass not greater than 500 kilograms (1100 pounds). The mass of the Model No. NAC-STC package (260,000 pounds) is greater than 500 kilograms. Therefore, the requirement to perform a crush test is not applicable to the Model No. NAC-STC package.

- Issue 11, Fissile Material Package Design for Transport by Aircraft.

  The final rule adopted a new section, Section 71.55(f), which addresses packaging design requirements for packages transporting fissile material by air. The applicant stated that this new rule is not applicable to the Model No. NAC-STC package. Therefore, the staff has revised the Certificate of Compliance to specify that air transport is not authorized.

- Issue 12, Special Package Authorizations.

  The final rule adopted provisions for special package authorization that will apply only in limited circumstances and only to one-time shipments of large components. This provision is not applicable to the Model No. NAC-STC package. Thus, no change is necessary to conform to the new rule.

- Issue 13, Expansion of Part 71 Quality Assurance (QA) Requirements to Certificate Holders.

  The final rule expanded the scope of Part 71 to apply to any person holding or applying for a Certificate of Compliance. QA requirements apply to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety. The applicant has a QA program meeting the requirements, as required for the Certificate Holder. The QA program has been approved by the NRC. Therefore, no change is necessary to conform to the new rule.

- Issue 14, Adoption of the American Society of Mechanical Engineers (ASME) code.

  This proposal was not adopted for the final rule. Thus, no change is needed to conform to the new rule.

- Issue 15, Change Authority for Dual-Purpose Package Certificate Holders.

  This proposal was not adopted for the final rule. Thus, no change is needed to conform to the new rule.

- Issue 16, Fissile Material Exemptions and General License Provisions.

  The final rule adopted various revisions to the fissile material exemptions and the general license provisions in Part 71 to facilitate effective and efficient regulations of the transport of small quantities of fissile material. The criticality safety of the NAC-STC package does not rely on limiting fissile materials to exempt or generally licensed quantities. Chapter 6 of the application demonstrates criticality safety of the package with the authorized fissile contents. Therefore, no change is necessary to conform to the new rule.

- Issue 17, Double Containment of Plutonium.

  The final rule removed the requirement that packages with plutonium in excess of 0.74 TBq (20 Curies) have a second separate inner container. The applicant revised the SAR to remove any text referring to double containment of plutonium for the Model No. NAC-STC package.

- Issue 18, Contamination Limits as Applied to Spent Fuel and High Level Waste Packages.

  This proposal was not adopted for the final rule. Thus, no change is needed to conform to the new rule.

- Issue 19, Modification of Events Reporting Requirements.

  The final rule adopted modified reporting requirements. The applicant indicated that no revisions to the application were required to address this change.

Based on the statements and representations in the application, the staff concluded that the design has been adequately described and meets the requirements of 10 CFR 71.19(e) for a "-96" package designation.

For the CY vacuum drying enhancements, the applicant revised Chapter 1 of the NAC-STC package safety analysis report (SAR) to incorporate the definition of the CY fuel inserts and intact fuel assembly, and the option of inserting solid stainless steel rods into CY intact and damaged fuel assembly reactor control cluster assembly (RCCA) guide tubes not containing RCCAs. As stated by the applicant, the changes are based on the previously approved MPC-03A and MPC-03 amendments, and are, therefore, acceptable.

2.0 STRUCTURAL

Evaluation

Attachment 2 to the August 4, 2005, letter lists the changes and their justifications for the 17 licensing drawings. The staff reviewed these changes and determined that they are generally administrative and editorial in nature in that minor revisions of non-safety related design/fabrication features were incorporated in the drawing updates. However, in Drawings 414-872 and 414-889 for the spent fuel and greater than Class C (GTCC) waste, respectively, the applicant changes the size of the structural lid-to-shell closure weld from 7/8" to 3/4" for the CY multiple purpose canister (CY-MPC). Similarly, the shield lid-to-shell weld is reduced from 1/2" to 3/8". To evaluate effects of the weld size changes on structural performance, the applicant used new weld dimensions in a finite element re-analysis of the canister. As reported in the revised stress summary tables, all calculated stresses remain to be below the allowables for the normal conditions of transport and hypothetical accident conditions.

In Drawings 423-800, 802, 803, 807, and 875 for the NAC-STC cask assembly, port cover assembly, and neutron absorber material, the applicant changed the material specifications on various parts and components (e.g., plug, relief valve, recess tubing, etc.). The applicant changed some components from ASTM/ASME specification to commercial, changed some to a different category within the same code, such as Parker designation, others from SS-304 to SS-316, and non-metallic O-ring materials to PTFE (polytetrafluoroethylene). Another example is changing the material of the stainless steel drain tube from ASTM code to commercial. In Drawing 423-811, the neutron shielding material is changed to solid synthetic polymer, NS-4-FR containing 0.6 wt.% B_4C. The staff reviewed these changes and determined that they are acceptable.

Conclusion

On the basis of the review above, the staff concludes that the changes requested do not affect the ability of the Model No. NAC-STC package to meet the requirements of 10 CFR Part 71.

3.0 THERMAL

The applicant's proposed change to allow the insertion of solid stainless steel rods into Connecticut Yankee intact and damaged fuel assembly RCCA guide tubes that do not contain an RCCA for spent fuel assemblies in the NAC-STC will have essentially no effect on the thermal performance of the cask. The applicant demonstrated that there would be no significant thermal impact from the insertion of as many as 20 rods into all fuel assemblies for any given cask loading. The assemblies that have stainless steel rods inserted in RCCA guide tubes are still bounded by the same parameters as the Approved Contents outlined in the original application (SAR, Revision 15, March 2004). The applicant has provided assurances in the SAR that the thermal models utilized in the original application bound the thermal response of Connecticut Yankee fuel with stainless steel rods inserted. Therefore all original thermal calculations and results apply.

The staff reviewed the applicant's evaluation and agrees with the conclusion that there are essentially no thermal effects from the addition of stainless steel rods to the Connecticut

Yankee fuel. The staff finds that the package meets the requirements of 10 CFR Part 71 for thermal performance.

4.0 CONTAINMENT

The applicant revised the containment analysis to calculate a new allowable leakage rate for the Model No. NAC-STC configuration with Viton O-rings, for use when the package is directly loaded for transport without interim storage. All other package configurations will continue to be tested according to the "leak-tight" definition of ANSI N14.5-1997, "Leakage Tests on Packages for Shipment."

The revised allowable leakage rate calculation was necessary to incorporate updated A_2 values in the latest revision of 10 CFR Part 71. None of the allowable contents changed, and the bounding assumptions used in the revised analysis are identical to those of the previously approved analysis. The resulting allowable leakage rate calculated for direct loading of the NAC-STC with Viton O-rings is 7.5×10^{-5} ref cm^3/sec, which corresponds to 9.3×10^{-5} cm^3/sec (helium).

The applicant has shown and the staff agrees that the Model No. NAC-STC, with the changes discussed above and in the SAR, continues to meet the containment requirements of 10 CFR Part 71.51(a).

5.0 SHIELDING

The applicant did not submit a revised shielding analysis for the changes requested in their amendment request. Changes to the shielding section of the SAR focused on removing all references to IAEA Safety Standard Series No. ST-1. The changes to the SAR also corrected the Transport Index from 20 to 21 in one location that is consistent with the previously approved revision. These changes were editorial in nature and do not affect the safety of the NAC-STC package.

The applicant also requested an addition to the contents of the NAC-STC cask system. This would allow the option of placing unirradiated stainless steel rods into the RCCA guide tubes for both intact and damaged fuel assemblies. The addition of these rods would tend to provide additional shielding in the lower regions of the cask and are therefore bounded by the previously approved analysis.

The staff reviewed the proposed changes and concluded that they do not affect the ability of the NAC-STC cask system to meet the radiation protection requirements of 10 CFR Part 71.

6.0 CRITICALITY

The applicant did not submit a revised criticality safety analysis for the changes requested in their amendment request. Changes to the criticality safety section of the SAR focused on removing all references to IAEA Safety Series No. 6. The changes to the SAR also changed the term, "nuclear criticality control transport index" to "Criticality Safety Index (CSI)" in several locations of the Criticality Safety section to comply with the requirements of 10 CFR 71.59.

These changes were editorial in nature and do not affect the criticality safety of the NAC-STC package.

The applicant also requested an addition to the contents of the NAC-STC cask system. This would allow the option of placing unirradiated stainless steel rods into the RCCA guide tubes for both intact and damaged fuel assemblies. A revised criticality safety analysis is unnecessary since the addition of these rods would be to reduce the effective multiplication factor due to both the water displacement and the additional neutron absorption from the added stainless steel rods and are therefore bounded by the previously approved analysis.

The staff reviewed the proposed changes and concluded that they do not affect the ability of the NAC-STC cask system to meet the criticality safety requirements of 10 CFR Part 71.

7.0 OPERATING PROCEDURES

The applicant revised the Operating Procedures for the Model No. NAC-STC package to change the vacuum drying procedure employed prior to sealing directly loaded fuel in the package for transport. This procedure was modified to increase the vacuum drying pressure from 3 mbar to 4 mbar. The applicant also revised the Operating Procedures to restrict the use of air during bulk water removal and cask drying operations.

The applicant also changed the Operating Procedures to incorporate the revised Viton O-ring leakage rate testing values for directly loaded fuel. These values were calculated in Section 4 of the SAR and confirmed by NRC staff as discussed in Section 4.0 of this SER.

The staff agrees that neither of these changes will significantly affect the ability of the Model No. NAC-STC package to meet the requirements of 10 CFR Part 71.

CONDITIONS

The following conditions in CoC No. 9235, Revision No. 9, have been revised as follows:

- Item No. 3(a) of the certificate was revised to correct the certificate holder's address.
- Item No. 3(b) of the certificate was revised to reference the supplements to the application.
- Condition No. 5(a)(3) of the certificate was revised to reference the updated packaging drawings.
- Condition No. 5(b)(1)(iii) of the certificate was revised to remove the reference to the fissile exemption limits of the previous version of 10 CFR Part 71.
- Condition No. 5(b)(1)(iv) of the certificate was revised to include stainless steel rods inside reactor control cluster assemblies (RCCA) guide tubes as authorized contents.

- Condition No. 5(c) was revised to replace the wording “Minimum criticality safety index to be shown on label for nuclear criticality control” with “Criticality Safety Index (CSI),” as defined in 10 CFR 71.4.

- Conditions 9(a) and (b) were revised to remove references to previous revisions of the NAC-STC SAR, which have been incorporated into the consolidated application.

- Condition No. 12 of the certificate was added to specify that air transport is not authorized.

- Condition No. 13 of the certificate allows a package to be marked with the previous package identification number, USA/9235/B(U)F-85 until April 30, 2007. This condition allows time to replace the packaging nameplate that shows the revised package identification number, USA/9235/B(U)F-96.

- Condition No. 14 of the certificate authorizes use of the previous revision of the certificate for a period of approximately one year.

- Condition No. 15 of the certificate was changed to clarify that the package is approved for use under the general license provisions of 10 CFR 71.17. This change is due to a revision in the numbering of the sections in 10 CFR Part 71, that became effective on October 1, 2004.

CONCLUSION

Based upon the staff's review, the statements and representations in the application, as supplemented, for the reasons stated in this Safety Evaluation Report, and with the conditions listed above, we conclude that these changes will not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9235, Revision No. 9,
on April 25, 2006.